APPENDIX A

REVIEW OF LICENSING/DESIGN, PROCESSES AND EVENTS THAT LED TO RADIOLOGICAL OCCURRENCES

1. LICENSEE'S HISTORICAL REVIEW

Scope

The licensee's preliminary historical review records and scoping survey plans were reviewed by the team to gain an understanding of the scope and extent of previous radiological occurrences at the Haddam Neck site. The NRC team developed information from licensee documentation of surveys and reports of spills and releases, as well as from results of licensee interviews with personnel. It is expected that this information will permit the NRC to better assess the licensee's site characterization and remediation efforts, and to determine the acceptability of the licensee's termination plan, as required by 10 CFR 50.82.

During the course of plant operation, radiological conditions developed in the Radiological Controlled Area (RCA) as the result of the processing and handling of radioactive waste and effluents. In some cases, areas in the RCA became contaminated. Though remedial action was taken by the licensee, residual contamination occasionally migrated from the RCA into the surrounding owner-controlled property. In other cases, events involving gaseous effluent releases may have deposited materials outside of the RCA. Also, the licensee's process for release of material from the RCA to unrestricted areas was not adequate. The review included a selected examination of the licensee's identification, assessment and follow-up actions for these situations.

1.1 Site Characterization

The purpose of site characterization is to identify the type, location and concentrations of contamination present on the Haddam Neck site in order to determine what remediation is necessary to decommission the facility. This information is used to estimate the volume and class of waste material, by evaluating the radioactive contamination of the land areas, systems and structures of the facility. Besides the decommissioning planning, it also supports the final status survey process by identifying the areas that may require more monitoring and sampling. NRC requires, through 10 CFR 50.75(g), that licensees keep records of information important to the safe and effective decommissioning of the facility in an identified location, until the license is terminated.

The regulation requires the licensee to maintain records of spills or unusual occurrences that result in significant contamination remaining after remediation efforts. In such cases, the licensee must implement adequate radiological controls to assure regulatory requirements are maintained. Provided that all regulatory requirements can be maintained, the licensee is not required by the regulations to fully remediate contaminated areas on its property to background levels. However, the contaminated areas must remain under the control of the licensee until released in accordance with regulatory requirements.

In early 1997, the licensee initiated a radiological characterization scoping survey which included survey and identification of potentially contaminated areas inside and outside the RCA. As part of this review, the licensee examined previous Adverse Condition Reports (ACRs), a problem identification and corrective action reporting system. On June 30, 1997, the licensee identified that the 10 CFR 50.75(g) decommissioning record file was not completely current and did not contain all the information required by 10 CFR 50.75(g). The licensee reported this discrepancy in ACR 97-0381. Subsequently, the licensee undertook an historical review to recreate this file, in conjunction with on-site scoping surveys, to identify the extent of on-site contamination in suspected contaminated systems and land areas.

The team noted that the licensee had initiated a 10 CFR 50.75(g) file in 1990 by compiling a list of previous events reported to NRC and from the licensee's Plant Incident Reporting System (now ACR system). However, as identified by the licensee in 1997, this file was incomplete and had not been maintained. The NRC inspection record does not indicate that the requirements of 10 CFR 50.75(g) were inspected. Further, areas outside the radiological controlled areas were not included in the licensee's file until they were identified by the scoping surveys conducted from July through October 1997. Such areas included the landfill area (shooting range); the hillside behind the fuel building (east side of the site); the storm drain run-off area south of the site; and the peninsula area (southwest storage area).

The licensee documented radiological surveys of the plant site starting in 1967. Dose rate surveys were performed quarterly for the perimeter and areas within the radiological controlled area boundary. In 1979, the licensee began an annual site survey which included areas outside the radiological controlled area boundary but within the owner controlled area. The surveys in 1979 included monitoring for loose (removable) contamination in addition to the dose rate measurements.

1.2 On-site Contamination

The licensee's site characterization effort also involved the identification of significant on-site contamination events. As of September 19, 1997, the licensee had documented or identified approximately 125 individual "events" (e.g., an activity, event or spill) that may have resulted in residual contamination of the site over its operating history (see Supplement A-1 to this Report). Of the 125 events, about 12 involved non-radiological type events (e.g., oil spill). The events, dating back through 1969, were documented in records such as abnormal occurrence reports, plant incident reports, licensee event reports, adverse condition reports and event notifications. In general, each event included an event description and a statement of what corrective action (including remediation efforts) was known to have been taken at that time. The licensee estimated that complete information (quantities of materials, drawings, documentation of remediation actions and survey records) was only available for approximately 10 percent of these events.

The licensee has performed (and is continuing to perform) radiation surveys of the site to document the type and levels of radioactive material present. One licensee report reviewed was the "Investigation of the Source of the Radioactive Contamination Found on the Connecticut Yankee Site March 10-30, 1980," dated April 1980. This report documented the results of extensive radiological surveys performed on plant buildings and site property. The surveys revealed the presence of licensed radioactive material in areas beyond the radiological controlled areas. The licensee identified and remediated areas where the radiation levels were above NRC limits for non-radiologically controlled areas. This information was reported to the NRC and the State of Connecticut Department of Environmental Protection within one hour of discovery. The licensee believed the likely source of the contamination resulted from the release of radioactive material through the primary vent stack after actuation of the degasifier rupture disk in February 1979, and again in December 1979, and possibly from residue from the cleaning of the stack in September 1979. The licensee performed a dose assessment which <u>assumed</u> the radioactive material was transported offsite and exposed a member of the public, and that the exposure was averaged over the entire skin of the whole body. The calculated potential doses to the skin and to the gastrointestinal tract were 0.75 mrem and 0.3 mrem, respectively. These calculated doses are a very small fraction of 10 CFR Part 20 annual dose limits and within the ALARA criteria of Appendix I to 10 CFR Part 50. However, the skin dose, when calculated over 1 square centimeter, which is consistent with regulatory guidance, could have approached the occupational quarterly limit for skin of the whole body (7.5 rem/quarter). Because these discrete spots of contamination were not widespread, the likelihood that a person would have received a skin contamination in excess of the occupational skin limit is remote.

The licensee’s report also identified residual levels of radioactive material in mud sediments along a storm drain runoff leading from the facility grounds to the discharge canal. The licensee identified that the contamination likely resulted from the discharge of contaminated liquid from the storm drain which originated within the radiologically controlled area and from runoff from the protected area. The contaminated runoff likely originated from contamination on the ground, which resulted from leaking radioactive liquid storage tanks and from radioactive waste handling operations in the outside environment but within the radiologically controlled area.

This radioactive material from the storm drain and the runoff represented an unmonitored release pathway. There was no barrier to prevent the radioactive material from migrating into the licensee’s discharge canal and being transported into the unrestricted area. Because the radioactive material released into the discharge canal through this pathway was not monitored, the licensee did not have data to support compliance with NRC regulations.

The licensee’s documentation indicated that areas of potential residual contamination were principally located around (and potentially under) various radiological controlled process buildings. A site map depicting suspected areas of residual contamination is included as Supplement A-2. These buildings are centrally located on the site and within the protected area. The licensee performed core bores at three locations around this area and noted that, based on these limited preliminary samples, no significant subsurface (e.g., greater than 6 inches) residual activity was present. Additional samples are planned.

Areas where residual contamination may be present included locations previously used for outdoor handling of radioactive waste (e.g., outdoor resin handling station). Other suspected locations include an area known as "the ballfield" (an area within the protected area) and an outfall area at the south end of the facility (outside the protected area but within the owner controlled area) known as the "leachfield."

The "ballfield" may have received potentially low-level contaminated fill soil from building excavation projects when it was paved over. The NRC team noted that an apparent excavation was performed to support the construction of the radwaste reduction facility, as described in plant design change request No. 85-733, dated October 15, 1985. In the course of the excavation, the licensee detected low-level soil contamination, excavated the contaminated soil to a pre-determined specific activity based on an evaluation, disposed of the material by transfer to a licensed disposal site and performed a dose calculation based on residual radioactivity remaining in the excavation. The licensee performed direct frisking of soil and analyzed it using gamma spectroscopy. Existing records indicated that any remaining contaminated soil was drummed and disposed of as radioactive waste. As discussed in Section 3 of this report, the licensee's release criteria were inappropriate, at that time, and may have resulted in the release of small concentrations of radioactive material.

In July 1997, the licensee became aware that an area known as the "landfill," located about 0.25 miles northeast of the station in the owner-controlled area, had received fill/rubble from a previous on-site work activity. The licensee performed radiological measurements at the location and detected low-level concentrations of radioactive material in the soil. Cobalt (Co-60) activity in isolated spots ranged from about 0.31 pCi/g to 4.3 pCi/g. Cesium (Cs-137) ranged from 0.17 pCi/g to 34.8 pCi/g. The licensee collected material from the area (e.g., fabric, soil, brick) that indicated 400 - 600 corrected counts per minute (ccpm). The licensee performed preliminary estimates of potential doses to personnel who may have inadvertently entered the area and concluded that any dose received would be well within NRC regulatory limits. The area was subsequently fenced in and designated as a radiological controlled area pending further evaluation.

The southwest site storage area (also known as the "boneyard") is located on the peninsula between the discharge canal and the Connecticut River. The area was used as a storage area for various items throughout the operation of the facility, including potentially radioactive/contaminated items. In addition, other portions of the peninsula were used for storage of dredged material from the discharge canal. As part of the site characterization, the licensee reviewed records to determine whether the area was surveyed periodically for radioactive materials. A survey performed by the licensee in March 1980 revealed a section of concrete slab with dose rates up to 500 millirem/hour on contact. The slab was buried under approximately 1.5 feet of soil. After removal of the concrete, there were no other areas found with elevated dose rates. A licensee investigation revealed that radioactive material had been stored in this location and the contamination could have been inadvertently left in the area when the material was removed.

Of the 125 radiological occurrences identified by the licensee, most did not result in any significant contamination. The following radiological occurrences resulted in some level of site contamination that may require further remediation to support decommissioning:

- Leak from the radioactive water storage tank (RWST) heater valve in November 1973 that contaminated the storm drain system;

- Multiple waste gas tank rupture disc actuations in the 1970s;

- Various leaks in the steam generator blowdown waste discharge line and the service water effluent line under the Primary Auxiliary Building (PAB) floor in the 1976-1980 time period;

- Contamination of the yard area around the Borated Water Storage Tank (BWST) from a leak in the circulating water heater line in 1978;

- Unplanned radioactive release from the degasifier through the plant stack in December 1979;

- Leak from a cracked weld seam in the auxiliary building exhaust duct to the main stack in September 1981;

- Resin liner overflows in 1984;

- Dredging of the discharge canal in 1986;

- Drain hose spill of contaminated water to the yard area in August 1987;

- Contaminated water from radioactive waste processing dumped into an uncontrolled drain that emptied into the 115 kV switchyard trench in February 1989;

- Spill of component cooling water to the storm sewer in March 1990;

- Leak from the Refueling Water Storage Tank in September 1990;

- Spill from the Reactor Coolant System to the pipe trench in August 1991; and

- Draining of the PAB heat exchanger to an uncontrolled drain that emptied into the 115 kV switchyard trench in April 1984.

Routine operations led to contamination of groundwater at the Haddam Neck site. However, the amount of contamination in groundwater leaving the site is limited by Appendix B of 10 CFR 20. NRC does not have a general regulatory position or guidance on groundwater monitoring at Part 30, 40, 50, and 70 non-waste disposal facilities. If any groundwater monitoring is performed at these sites, it is done through license specific requirements. Although not specifically provided to reactor licensees, the NRC's Nuclear Material Safeguards and Safety (NMSS) Low Level Waste and Decommissioning Projects (LLDP) Branch did publish, and notice in the Federal Register, a Branch Technical Position (BTP) entitled "When To Remediate Inadvertent Contamination of the Terrestrial Environment" in October 1994. This BTP recommends to licensees that known or suspected releases to groundwater need to be characterized, and remediated as appropriate, as soon as possible.

Timely remediation would minimize health and safety problems. The continued presence and movement of contaminated soil and/or groundwater over time could also increase the volume of contaminated material and therefore increase the cost of decommissioning. However, by regulation, power reactor licensees are not required to remediate areas that are inaccessible until decommissioning.

Tritium from routine effluents and spills is present in the groundwater on-site. Tritium is highly mobile in the environment and is easily detected in groundwater samples after a release. The groundwater contamination at Haddam Neck was monitored in the radiologically controlled area at the external containment sump (ECS) near the containment building, and outside the radiologically controlled area, but on the owner-controlled property, at two water supply wells adjacent to the discharge canal. The primary source of the tritium was identified as coming from waste test tanks. The Haddam Neck staff first identified the source of the contamination in the sump in 1976. The source of the tritium in the ECS was suspected to be due to leakage from monitoring tanks. The source of the tritium in the wells was suspected to be due to migration of tritiated water in the discharge canal to the local aquifer penetrated by the wells. Because the water from the two wells was a nonpotable water source for the facility, with tritium concentrations above background, this could have been an unmonitored dose pathway, and it might not have been within the principles of As Low As Reasonably Achievable (ALARA) to use this water for any domestic purposes at the site. Although the on-site well water was used for process water, there were no controls to prevent facility workers from drinking the tap water. However, the dose calculations in NRC Inspection report 50-213/97-11 indicate that the potential doses from tritium, even if the water was used as a drinking water source, would have been low (< 1 mrem/yr) and not a health and safety concern.

EPA Interim Drinking Water regulations in 1976 established a maximum contaminant level (MCL) of 20,000 pCi/l for tritium. These were written for public drinking water supplies serving 25 or more people. The drinking water MCL does not apply to Haddam Neck's use of the groundwater. EPA's CERCLA program guidance requires the application of MCLs in the groundwater plume that is a current or potential source of drinking water. In addition, the NRC's 1992 Site Decommissioning Management Plan (SDMP) Action Plan for decommissioning suggests that MCLs be used as reference standards for groundwater remediation at decommissioning sites. However, the recently promulgated decommissioning criteria rule specifically excludes using MCLs as a separate standard for groundwater contamination at decommissioning sites. NRC is aware of several NRC licensees that have contaminated groundwater on-site. The fact that a licensee has contaminated the groundwater at its site (above MCLs in some cases) is not a specific violation of NRC regulations unless the regulatory requirements of 10 CFR 20 or license conditions are exceeded. Notwithstanding, the potential dose from any groundwater radioactive plumes would be determined during site characterization. The evaluation would determine if groundwater remediation would be required.

Haddam Neck staff kept records of the groundwater tritium concentrations over the years, and while the source of the contamination in the sump was believed to be known and remediated, there continued to be tritium in the sump water at varying concentrations. This is somewhat problematic as other sources of tritium may have been contributing to the concentrations found in the sump over the years, but their presence would have been masked by the previous contamination and the assumption that the previous contamination was the source of all subsequent tritium levels measured in the sump. Apparently, no attempt was made to characterize the groundwater plume that was contributing tritium to the sump. It is possible tritium in the groundwater could be coming from an unidentified source. In addition, without characterizing the plume there is the potential that the measuring point(s) are not in the correct location to detect the maximum concentration of the plume.

Groundwater at Haddam Neck flows into the Connecticut River, which is not a drinking water source, downstream of the site. Thus, dose to the pubic via the drinking water pathway is essentially zero.

Conclusion

The scope and depth of the licensee's current effort to review and document the site's history regarding contamination events and radiological occurrences are appropriate and sufficiently comprehensive. The licensee has identified over 125 events, some of which contributed to the current radiological condition of the facility, that could have an impact on the decommissioning. While these events resulted in the potential for, or the occurrence of, radioactive materials being released outside the confines of the RCA, the licensee's radiation survey of the site and evaluation does not reveal any evidence that the quantities or concentrations represented a significant radiological hazard to plant workers, members of the public or the environment. Where the radiation levels exceeded NRC regulations or reporting criteria, the licensee made the appropriate reports and remediated the areas.

NRC regulation 10 CFR 50.75(g) requires recordkeeping of spills or other unusual occurrences involving the spread of contamination in and around the site. However, the records may be limited to instances when significant contamination remains after any cleanup is done. The regulation does not require remediation to background radiation levels. The records of the contamination and its location will be used to decommission the site. Prior to the 1990 effective date of the regulation, the licensee was not required to have specific records on contaminated areas to facilitate the ultimate decommissioning, except for records related to on-site waste disposals. Routine surveys of the radiological controlled areas of the facility would have been performed to demonstrate compliance with the radiation standards in 10 CFR Part 20. For the significant contaminated areas identified by the licensee at Haddam Neck, complete records in accordance with 10 CFR 50.75(g) were not available to the team during the review. NRC Inspection Report No. 50-213/97-08 identified this as an unresolved item.

The tritium concentrations, although below the MCLs, are an indication of previous or current leakage from systems that contain radioactive materials. Tritium is highly mobile in the environment, so it is easily detected in groundwater after a release. Other non-soluble radioactive contaminates would not normally be expected to be detected in groundwater. The tritium monitored at Haddam Neck could indicate that some soil near the original spill or release point may require remediation. However, dose to the public via the drinking water pathway is essentially zero. Characterization of the tritium plume has been initiated as part of the licensee's site characterization.

2. OFF-SITE CONTAMINATION

Scope

This section reviewed the recent efforts by the licensee to characterize the quantity and concentration of materials that were released from the site. The licensee's offsite characterization has not been completed.

General requirements for disposition of licensed materials are listed in 10 CFR 20.2001. Because this regulation does not define an exempt quantity, any amount of detected licensed material must be dispositioned in accordance with NRC requirements. By using sensitive laboratory methods, trace amounts of licensed material may be detected at levels considerably less than the Lower Limit of Detection specified through other regulatory guidance or requirements. At such levels, there is no expected impact on public health and safety.

Details

2.1 Offsite Soil Releases

Various property owners informed the licensee that they had received soil from the plant, along with general fill material (asphalt, concrete, soil) during plant construction projects in the 1980s and 1990s. The fill was excavated from the site when CYAPCo constructed new buildings on-site (such as the emergency operations facility, the switchgear building and the radwaste reduction facility) and renovated a parking lot on the north side of the site. Although most excavated materials were taken to the licensee's landfill area on the south side of the site, a considerable amount of material also was released to the public for unrestricted use. The fill materials came from both inside and outside the radiological control area at the plant.

The licensee identified about 12 offsite areas that were believed, with reasonable assurance, to have received some fill/rubble from the site. These areas were identified by direct contact with various local property owners and by public response to notifications, press releases and media reports on the matter. Areas to be investigated were assigned to a matrix to positively identify the area for follow-up and to develop information on the time and circumstances under which the materials were received. The licensee initiated a walkdown of the subject properties to identify the areas potentially affected by plant-related materials. The results of the site walkdown were used to develop a specific survey and soil sampling plan of the suspect areas at each location.

The licensee also conducted similar surveys and soil sampling of areas on-site that were open for unrestricted public access, such as the north parking lot, the picnic areas, the boat launch access area and the nature trails on the north and east side of the plant.

2.2 Contaminated Blocks from Shield Wall

As part of the decommissioning process, the licensee's historical site survey identified that material from radiologically suspect areas of the plant had been released off-site. Specifically, based on interviews with plant workers who received the material, the licensee identified that workers took possession of solid cement blocks following the demolition of a wall in 1975. The blocks had been used as a shield wall around a former cask washdown pad that presently is the location of Bus 1O. The licensee estimated that 5,130 blocks were used to construct the wall. The blocks measure 4" X 8" X 16".

In the early 1970s, the cask pad was used for temporary storage of contaminated filters, resin liners and trash. At least one liner leaked. The leakage contaminated the storage area, including some of the blocks which contained it. Once the failed liner began to dry, airborne radioactivity was identified in the area. One worker recalled that remnants of the failed liner had contact levels of 10 R/hr.

After abandoning use of the cask pad as a storage area, the licensee dismantled the wall and began a process to survey the blocks to separate the contaminated ones from those unaffected by the contamination, with the intent to release the uncontaminated blocks. Plant workers were allowed to take blocks directly from the partially dismantled shield wall and to frisk the blocks for free release. When interviewed in 1997, most workers did not remember the type of survey instrument(s) used, or the release criteria that applied. While workers stated they checked the blocks for radioactivity, it was not certain that every worker checked each block. Health physics technicians helped some workers check blocks for contamination. Some workers, who were qualified in radiological controls, surveyed blocks during work shifts and separated blocks to be released into piles for each worker. The workers loaded the blocks into a truck at the end of a shift and removed them from the site. Based on entries in a security gate log, the process of frisking and taking possession of blocks occurred over the period of September through November 1975. Several workers took many truckloads of blocks. Subsequently, the blocks were used to build structures, walkways, ramps, retaining walls and landscaping borders. Some blocks were used inside the home (i.e., cellar).

In late 1997, the licensee issued a Licensee Event Report (LER) to the NRC regarding the breakdown in the health physics program that led to the release of contaminated material from the site. The licensee initiated corrective actions, including the survey, evaluation and removal or disposal of contaminated materials. NRC inspectors have performed independent measurements and analyzed split samples with the State of Connecticut and the licensee. The preliminary results from the NRC analysis of these samples indicate agreement with the licensee data. The dose assessment from the preliminary dose rate survey indicates the dose to a member of the public from contaminated soil is approximately 1 millirem per year. The highest dose rate from licensed material found off-site was less than 2 millirem per hour on contact, although the dose rate decreased substantially at a distance of 10 cm. As material was located, during the licensee's initial scoping surveys, the locations were remediated to less than 0.5 millirem per hour and 10 millirem per year. Final remediation criteria have not yet been established.

Conclusions

The scope and depth of the licensee's current effort to review past radiological occurrences and assess significance are appropriate and sufficiently comprehensive for the site radiological characterization, as required by 10 CFR 50.82(a)(9)(ii). Areas have been identified, both on and off the site, that have measurable radioactive contamination that may require remediation. However, the maximum dose to an individual, including members of the public, from this contaminated material, in the locations examined to date, is below the regulatory limits in 10 CFR 20. There is a potential that the location or use of some of the material may have resulted in higher doses in the past. The final determination of this matter will require additional assessment by the licensee.

There were no records of surveys for excavation of soil outside the RCA. The licensee has recently sampled, analyzed and reported the results of contamination in the areas beyond the radiologically controlled areas. The radioactive material was not quantified and evaluated prior to being released in order to determine if it represented a significant pathway that should have been controlled and monitored in accordance with NRC regulations. However, any residual contamination on the site will be identified during the licensee's current site characterization program and will be evaluated for compliance with the decommissioning regulations for license termination.

The licensee is currently performing a full-scope radiological characterization of the site in order to safely decommission the facility. Continuing NRC inspections will monitor the licensee's regulatory compliance with the regulations.

3. LICENSEE MATERIAL RELEASE PROCESS

Scope

A selected review was performed of the licensee's procedures for conducting radiation surveys of materials to be released for unconditional use. The procedures required surveys and/or evaluations in accordance with 10 CFR Part 20 to ensure that licensed radioactive material was not inappropriately released. The review compared the instructions in the licensee's procedures against the guidance contained in NRC Information Notices, regarding what constituted a reasonable survey/evaluation.

Details

The earliest procedures available to the team were Standardized Procedure #17, Unconditional Radiological Release of Material Offsite, Revision 0, dated October 20, 1981, RAP 6.2-14, Unconditional Radiological Release of Material Offsite, Revision 0, dated January 28, 1982, and RPM 2.2-8, Unconditional Release Surveys, Revision 0, dated

January 13, 1989. These procedures described the means by which material, that could potentially be contaminated, must be surveyed prior to being unconditionally released from the radiation controlled area.

The 1981 procedure did not contain release criteria guidance written in a practical format for use by the technician performing the survey. It appears that the procedure was quickly (within three months) superseded by RPM 2.2-8. The procedures from 1982 and 1989 provide specific instructions for the radiation survey of solid materials that may have fixed and/or removable surface contamination. The procedures specify that material containing detectable radioactive material, defined as 100 counts per minute above background, for beta-gamma surveys and 4 counts per minute above background for alpha particle surveys, is not to be released for unconditional release. The procedural guidance is consistent with early 1980s industry practices and NRC guidance published in Information Notice 81-07, "Control of Radioactive Contaminated Material (5/81)."

The information notice discussed that licensees are to perform adequate radiation surveys of waste with the potential to be contaminated with licensed material to ensure that licensed radioactive material is not inadvertently released. However, the notice specifically recognized that there would be levels of licensed radioactive material that could not be detected with commonly used radiation detection instruments and would be released into the general environment. The notice provided guidance on the minimum acceptable radiation detection capabilities for commonly used survey equipment; but, it did not provide release limits for radioactive material. The notice also acknowledged that there are other more sensitive analytical capabilities available to distinguish very low levels of radioactive contamination, noting that those capabilities are very elaborate, costly and time-consuming, making their use impractical (and unnecessary) for routine operations. Further, the notice stated that, based on the specified minimum detection capability, the potential radiation dose to members of the public from the release of any undetected, uncontrolled contamination would be significantly less than 5 mrem per year. This was considered by the NRC to be an acceptable dose criterion in 1981, since it was well below the explicit public dose limit of 500 mrem in 10 CFR Part 20. The industry generally viewed this information on required minimum detection capability for surveys as release limits. The NRC viewed licensee procedures that used the guidance in Information Notice 81-07 as acceptable.

In 1985, the NRC updated its radiological survey guidance for the unconditional release of potentially contaminated material to reflect the growing concern about the inadvertent release of licensed radioactive material. The updated guidance, which addressed the need for licensees to perform more sensitive surveys for large surfaces and packages of aggregated wastes, was published in Information Notice 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities (12/85)." The licensee's procedure, which was written in 1987 and referenced IN 85-92, did not address the updated NRC guidance. The absence of this updated and improved survey guidance in the licensee's procedure is not indicative of a good survey program for detecting surface contamination but not contrary to 10 CFR Part 20.

With respect to surveys for volumetric materials, NRC did not provide survey guidance or establish a release criteria for residual contamination. However, the licensee's procedure from 1982 had used an acceptance criteria for release if both of the following were shown by isotopic analysis:

> "Each isotope present does not exceed the exempt concentration specified in 10 CFR 30.70 Schedule A and that the sum of the isotope fractions is equal to or less than unity, and
>
> The total amount of each isotope present is not greater than the exempt quantity specified in [10 CFR 30.71] Schedule B."

Further, licensee records of surveys performed in the early 1980s denoted the inappropriate use of the values in 10 CFR Part 30, Schedule A, exemption tables. It appears that the licensee used these values as release limits, which is contrary to the requirements of 10 CFR Part 20. The dose impact from using the concentrations above the effluent release limits has not been evaluated because of insufficient information. The team believes that radioactive material in the public domain that has already been identified must be assessed by the licensee for the potential dose to the public.

For the radiation survey of a liquid or granular solid that may contain licensed radioactive material, the licensee's 1987 and 1989 procedures required that a representative sample of the material be counted on a gamma-ray spectrometry system. The system's lower limit of detection (LLD) for radioactive material is reported to be 3E-6 µCi/ml. The procedure states that this LLD corresponds to the most restrictive radioactivity concentration value in Appendix B, Table II, Column 2 of 10 CFR Part 20. It is further stated that, in practice, the system will be able to achieve a lower LLD than 3E-6 µCi/ml. Survey records from a December 1985 period that documented the survey of dirt in the RCA, using hand-held instruments, indicated that the licensee used a release limit of 1000 dpm/100 cm^2 and gamma analysis on a limited number of soil samples. Survey grid plans for the sampling size were not evident. Positive identification of licensed radioactive material was not acknowledged if the reported survey value was less than the isotope's concentration value in Schedule A, of 10 CFR Part 30 - exempt concentrations. For Co-60, a measurement result below 5E-4 µCi/ml[2] was apparently considered exempt based on the December 1985 survey record. This practice established release limits for radioactive material contained in solids intended for release to unrestricted areas and is contrary to 10 CFR Part 20, which only allows licensed radioactive material to be disposed of in specifically described ways.

Licensed radioactive material can only be disposed of in accordance with the methods described in 10 CFR Part 20. All other material that may be potentially contaminated with licensed material must receive a radiation survey. If any licensed radioactive material is detected, the material must be handled in accordance with 10 CFR Part 20. For the types of material and radionuclides typically observed at nuclear power plants, there are no release limits for detectable quantities of radioactive material contained in solid form released to unrestricted areas. Notwithstanding, the application of these procedures, containing inappropriate guidance, permitted the licensee to release solid materials that may have contained detectable quantities of radioactive materials, contrary to the requirements of 10 CFR 20.

For the radiation survey of a liquid or granular solid, the licensee’s use of an LLD of 3E-6 µCi/ml and the exemption schedules from 10 CFR Part 30 were not consistent with NRC guidance published in Information Notice 88-22, “Disposal of Sludge from On-site Sewage Treatment Facilities at Nuclear Power Stations (5/88),” and is contrary to 10 CFR Part 20. This Information Notice discussed the need for licensees to perform radiation surveys of representative samples of material under conditions that provide an LLD appropriate to measurements of environmental samples. Such measurements make it possible to distinguish licensed radioactive material from natural and fallout radioactive material. For the analysis of Co-60, the appropriate LLD for environmental samples of dry sediment is 1.5E-7 µCi/ml and 1.5E-8 µCi/ml for water samples. Thus, the licensee’s survey program for liquids and solids was not able to adequately detect small quantities of licensed radioactive material within bulk quantities of liquid or granular solid material being released for unrestricted use. The licensee’s stated LLDs were generally acceptable for use during the early 1980s, because the available gamma spectrometry systems at that time were not able to routinely achieve the low LLDs. Only expensive state-of-the-art systems could achieve the environmental LLDs. The NRC did not require licensees to have such sophisticated systems at that time. Consequently, licensees typically sent their environmental samples to a contractor laboratory for analysis, while surveys of bulk material for unrestricted release used less sensitive LLDs. However, since the early to mid-1990s, gamma-ray spectrometry systems that readily achieve the low LLDs are readily available at a reasonable cost. These systems are now routinely being used in the majority of nuclear power plants for routine use (i.e., release surveys of material).

The team reviewed the licensee’s current procedures for the survey and release of material, RPM 2.6-16, Revision 7, dated 10/22/97 and RPM 2.2-22, Revision 0, dated 8/21/97. These procedures contained updated survey guidance which used more sophisticated equipment and techniques. For liquids and granular solids, the licensee’s procedure required that the radiation surveys be performed to LLDs that are consistent with the environmental monitoring program. The licensee’s procedures are generally consistent with current industry practices, NRC guidance, and 10 CFR Part 20. Other than using the dates on the licensee’s procedures, the inspectors were not able to determine when the licensee updated survey criteria to use the environmental LLDs. Based on the guidance in RPM 2.2-8, the licensee was able to count lower than the stated LLD of 3E-6 µCi/ml. Thus, it appears that the licensee may have made a gradual transition to the use of the environmental LLDs over the years as new radiation detection equipment was installed.

Conclusions

Through 1989, the licensee’s material release process for removable and surface contamination was generally consistent with NRC criteria and industry practices. However, it did not contain appropriate criteria for surveys of volumetric materials (i.e., soil, sludge and debris). Additionally, the licensee did not keep up with improvements within the industry to increase the sensitivity of radiation surveys. This deficiency was observed in the licensee’s procedures, which had not incorporated updated NRC guidance for survey and release criteria from 1985 through 1989. The licensee’s use of the exemption schedules from 10 CFR Part 30, Schedule A, in its survey and release procedures as a release criteria was not appropriate and contrary to the requirements of 10 CFR Part 20. Further, the use of the annual liquid effluent release concentration contained in Appendix B to 10 CFR Part 20 was also contrary to the requirements of 10 CFR Part 20. As a result, the licensee's survey procedures were not adequate to prevent the release of licensed radioactive material from the site and is contrary to 10 CFR Part 20. The licensee records show that radiation surveys were generally performed on most of the material released from the radiological control area of the site, in accordance with the procedures. Use of the inappropriate criteria resulted in radioactive material being inadvertently released from the controlled areas at concentrations above effluent release limits.

Dose assessments from prior release of material with residual contamination have not been completed. Dose estimates from recently identified materials in the public domain are well within the NRC annual exposure limits specified in 10 CFR Part 20. The team recognizes that the breakdown of the radiation protection program in 1975, which caused the release of the concrete blocks without an appropriate survey, could have resulted in exposures to the public in excess of 10 CFR Part 20 limits. However, preliminary assessments of the as found use and condition of the blocks (e.g., walkways, garden borders, foundation supports) have shown potential dose impact to the public to be negligible, to date. The licensee's current procedures for the survey and release of materials are consistent with current NRC guidance and 10 CFR Part 20.

4. RADIOLOGICAL ENVIRONMENTAL MONITORING REPORTS

Scope

A selected review was performed of the licensee's Annual Radiological Environmental Operating Reports to determine if licensed radioactive material of plant origin was observed in the environment outside the plant site. The licensee's reports were also reviewed for compliance with its radiological environmental monitoring program.

Details

NRC regulations require licensees to keep levels of radioactive material in effluents ALARA (as specified in 10 CFR 50.34a) to ensure that radiation doses to the public resulting from effluent releases or other radioactive material of plant origin will continue to remain minimal. To verify whether exposures in the environment are within the limits of 10 CFR Part 20 and to ensure that there is no long-term buildup of specific radionuclides in the environment, NRC requires licensees to monitor the environment for radioactivity of plant origin. This requirement is contained in General Design Criterion 64, "Monitoring Radioactivity Releases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities." The licensee's radiological environmental monitoring program is designed around the NRC's requirements to establish correlations between levels of radiation and radioactivity in the environment and radioactive releases from the plant, as well as, to provide supporting evidence that the impact on the environment from plant operation is within the analysis contained in the plant's licensing basis documents (i.e., the Final Environmental Statement).

The review compared NRC regulations and regulatory guidance against selected examples of the licensee's Annual Radiological Environmental Operating Reports from 1979 to 1996. The licensee's reports contained adequate documentation of the required sampling, analyses, interpretations and discussion of results, historical trends, land use census, Quality Assurance program and a discussion of calculated dose commitments to a member of the public. The reports contained a discussion of the calculated radiation dose to a member of the public based on two methods: a calculation based on monitored radioactive effluents released into the environment and on calculations based on the concentration of licensed radioactive material observed in environmental media (fish, milk, vegetation, etc). The licensee reported that measurable levels of radioactive material, attributed to plant operation, were observed in selected environmental media. With the exception of tritium, all the reported concentrations of observed licensed material were within regulatory requirements and did not require a special report. As noted by the licensee, significantly higher levels of tritium than background were detected and reported. However, the calculated dose consequences to a member of the public from the radioactive material were within the regulatory requirements of 10 CFR 20 and Appendix I to 10 CFR 50.

The licensee's Annual Radiological Environmental Operating Reports discussed that elevated tritium concentrations were routinely observed in water samples obtained from the on-site indicator well. The licensee's reports noted the elevated levels and acknowledged that the tritium was a product of plant operation. The licensee further explained that the tritium in this well water was within an area influenced by radioactive effluents released in the discharge canal and that tritium has the capability to readily follow the flow of groundwater. The flow of this ground water is to the Connecticut river. The licensee states that the tritium in the groundwater and the river water has no dose consequence on the public, or plant workers, since the water is not used for drinking. Tritium was also observed above background levels in samples from the Connecticut River, but, at concentrations significantly lower than the samples from the on-site well.

The information contained in the licensee's reports was consistent with the guidance contained in Regulatory Guides 4.2 and 4.8 and Criterion 64 of Appendix A of 10 CFR Part 50. This regulatory guidance has remained essentially unchanged since it was introduced in the early 1970s.

In addition to the licensee's environmental monitoring program, the State of Connecticut performed independent environmental monitoring around the site. The state's program was partially funded by the NRC. The purpose of the state program is to obtain environmental monitoring data that is independent of the licensee's data. The state collects samples of environmental media from the same locations as the licensee and independently analyzes the samples. The results are reported to the NRC in an annual report. The review team examined the reports that were readily available for the years 1994 through 1996. The state reported "substantial agreement" between their data and the licensee's. No unusual conditions or levels of radioactive material were noted. The team noted that as of 1998, the NRC no longer provides funding to the states for independent environmental monitoring.

Conclusions

Overall, the radiological environmental monitoring data contained in the licensee's reports were developed in accordance with regulatory guidance, were properly documented, and were reported in accordance with Technical Specifications and regulatory requirements. No errors or omissions were identified. The licensee's radiological environmental monitoring program adequately established correlations between levels of radiation and radioactivity in the environment and radioactive releases from plant operation. It provided supporting evidence that the impact on the environment from plant operation is within the analysis contained in the plant's licensing basis documents (i.e., the Final Environmental Statement) and 10 CFR Part 20. No significant adverse environmental impacts were observed by the licensee's environmental monitoring program as a result of routine effluent discharges or from radioactive contamination that originated from the plant's RCA. The licensee's documentation is consistent with the findings of the Final Environmental Statement issued by the Atomic Energy Commission in October 1973. The review team did not identify any areas of the licensee's program, beyond those already identified in NRC inspection reports that were in violation of NRC regulations.

5. RADIOACTIVE WASTE SYSTEM

5.1 Licensing Design Basis

Scope

The review consisted of examination of documents from the issuance of the 1966 Haddam Neck Facility Description and Safety Analysis (FDSA) to the 1987 issuance of the Updated Final Safety Analysis (UFSAR). Documents reviewed included pre-operational safety analyses, the Provisional Operating License and amendments, the Full Term Operating License and amendments, the Final Environmental Statement, Facility Description and Safety Analysis (FDSA), Updated Final Safety Analysis Report (UFSAR) and plant design change requests related to radwaste systems. Documents were examined on site at Haddam Neck, in the NRC Region I offices and at NRC headquarters.

Background

The licensing basis of Haddam Neck's radwaste processing systems was examined to determine whether the location and use of the major systems were within the licensing basis. Other issues addressed included the extent to which spills of radioactive materials may be within the licensing basis and the interaction of fuel cladding defects on the design basis of radwaste processing systems. The adequacy of the installation of an extensive modification of the radwaste processing systems completed in 1975 was considered.

The licensing basis includes NRC regulations, the license, orders, exemptions, technical specifications, the Final Safety Analysis Report (FSAR) and plant modifications, among other documents. Because the regulations, licensing documents and the plant itself changed over time, the licensing basis also changed. In addition, the understanding of what constitutes the licensing basis has changed over time by widening what was included in the definition. The overall effect of the changes has been to increase the margin of safety associated with nuclear power plant operation and to provide greater quality assurance through more extensive documentation.

The licensing basis defines the design and operation of a nuclear facility to provide several layers of defense-in-depth protection of the public health and safety. The health effects of radiation have been well studied. Accordingly, regulatory limits are established well below levels that cause harm, so that operation of a nuclear power plant within regulatory limits will cause no significant public health and safety effects. To assure that regulatory limits are not exceeded, the licensing basis adds a margin of safety by establishing safety limits that are more conservative than the regulatory limits. The safety limits include surveillance requirements so that the licensee will observe the condition of the plant and take corrective action in a timely manner. Sound design and high quality established by the licensing basis minimizes the possibility that malfunctions can occur. However, the plant design includes provisions, such as requiring multiple systems to perform important functions, to safely contain radioactive materials even if some equipment does malfunction or if a mistake is made. If multiple system failures should nevertheless occur, emergency procedures provide methods to mitigate the consequences of an accident and protect the public.

The defense-in-depth philosophy has been successful in preventing any significant public health and safety effects due to the operation of nuclear power plants in the United States.

<u>Details</u>

Connecticut Yankee Atomic Power Company filed its Facility Description and Safety Analysis (FDSA) for the Haddam Neck nuclear plant on July 19, 1966. Although not specifically mentioned, it is clear from the descriptions and knowledge of the plant layout that some waste handling would necessarily have to occur outside of buildings and enclosures.

The FDSA notes that the design basis of the radwaste systems included the assumption that 1% of the fuel fission products would be released into the reactor coolant by diffusion out of the fuel pellets and through cladding defects. The gaseous waste treatment system design used a somewhat different criterion, by addressing the magnitude of potential releases due to defects in 1% of the fuel rods.

Tritium was addressed in the FDSA by assuming 50% of its production would be released into the reactor coolant. Calculations demonstrated that if all the expected tritium inventory in the reactor coolant system (4015 Ci/yr) was released to the environment, the average concentration in effluents would still meet 10 CFR 20 limits.

The U.S. Atomic Energy Commission (AEC) issued a safety evaluation in connection with the plant's proposed Provisional Operating License on May 12, 1967. AEC accepted the design basis of defects in 1% of the fuel. The staff concluded that normal operation within the limits of the technical specifications would not result in potential offsite exposures in excess of 10 CFR 20 limits. The original technical specifications did not contain a fuel cladding defect limit.

The 1967 safety evaluation noted that the storage and hold-up facilities were located "outside containment" and analyzed the consequences of a waste gas sphere rupture. The dose at the site boundary was found to be within 10 CFR 20 limits. It was also noted that the Connecticut River was not used for drinking water supplies downstream from the site. Therefore, an accidental release of radioactive liquids into the river from the plant would not affect public water supplies.

Haddam Neck received its Provisional Operating License on June 30, 1967, which authorized operation at approximately 80% power. The plant began commercial operation on January 1, 1968. Full power operation was authorized on March 3, 1969.

AEC issued a safety evaluation on July 1, 1971, in response to the licensee's request for a full-term operating license. The SER noted that subsequent to the issuance of Haddam Neck's Provisional Operating License, the Commission had published General Design Criteria (GDC), effective May 21, 1971. The staff found that Haddam Neck conformed to the intent of the GDC. A design change to the radwaste systems was noted. The change allowed use of demineralizers in place of the originally installed aerated liquid waste evaporator, which had not met performance expectations, and accommodated liquid waste flow rates that exceeded the original design values. The staff concluded the design change met the ALARA criteria. The overall conclusion was that continued operation of the Haddam Neck plant would not endanger the public health and safety.

By 1972, the licensee was aware that its existing radwaste systems would not meet the requirements of proposed 10 CFR Part 50, Appendix I. An extensive modification was initiated in 1972, and made operational in 1975, to meet the new limits. The modification relocated the waste gas sphere from an outside area to a newly built waste-processing building. In addition, an evaporator was added to process waste liquids.

In December 1972, the licensee sent the design of the radwaste system modification to AEC. The licensee committed to issue an amendment to its license application upon completion of the modification. AEC acknowledged receipt of the design. However, no record can be found to demonstrate that the licensee submitted a license application amendment. The FSAR, reissued on October 15, 1975, and last updated in 1981, described only the original plant radwaste treatment equipment, not the modified system which had been operating since July 1, 1975.[1] A description of the modification was later included in the 1987 issuance of the UFSAR.

AEC completed the Final Environmental Statement for Haddam Neck in October 1973. The environmental impact of the plant as it existed at the time was evaluated and found not to endanger the public health and safety. Among other items, the FES considered the release of tritium to the environment. AEC estimated that all tritium produced in the core (8000 Ci/yr) could be released without exceeding regulatory limits.

AEC evaluated the expected performance of proposed radwaste systems modifications on normal effluents in the October 1973 Final Environmental Statement (FES). The FES contained simplified diagrams of the anticipated changes. The FES concluded that dose to individuals was within design objectives and ALARA. Dose to the population in the 50-mile radius was a small increment of natural background fluctuation, considered to be immeasurable and constituting no meaningful risk. The calculated population dose was lower for the modified radwaste system design than the original design.

The October 1973 FES considered the radiological impacts of a series of postulated accidents using the proposed guidance of 10 CFR 50 Appendix D, Implementation of NEPA. (Appendix D was revoked when 10 CFR Part 51 incorporated NEPA requirements.) The basis of the evaluation was the original plant design. Included in the consideration were Category 2 accidents, accidental spills and releases of radioactive materials outside containment, including those due to such developments as relief valve actuation. Doses were found to be within 10 CFR 20 limits for this category. Accidents analyzed in Category 3, Radwaste System Failures, included analysis of a rupture of the waste gas decay sphere specified in the original plant design. The consequences were within 10 CFR 20 limits, assuming an operable radiation monitoring system and that the licensee took some mitigating actions. Whole body dose for that accident was calculated as 0.185 rem at site boundary. Category 5 accidents involved release of fission products to primary and

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1. Haddam Neck was not required by regulation to make periodic updates to its FSAR prior to 1987. NRC considered the need to require periodic updating of the Final Safety Analysis Report in proposed rulemaking published for comment on November 8, 1976 (41 FR 49123). At the time, there was no requirement for a licensee to incorporate revisions, changes or amendments into the FSAR except where a hearing was held on an operating license application. After Haddam Neck received its FTOL in 1974, no updates to the FSAR were required until the FSAR updating rule became effective July 22, 1980 (45 FR 30614). However, Haddam Neck was exempted from the rule due to its participation in the SEP.

NRC announced the Systematic Evaluation Program (SEP) for a number of older plants in November 1977. The objective was to determine and document the degree to which they met licensing requirements for new plants. Haddam Neck was among those affected. As a result, Haddam Neck was exempted from the requirements of 10 CFR 50.71(e) to update its FSAR until after the SEP was complete. A number of extensions to the completion date was issued such that the licensee was not required to submit an updated FSAR until June 30, 1987. Haddam Neck made the submittal on June 22, 1987.

secondary systems due to fuel cladding defects, primary to secondary leakage and steam generator tube rupture. The consequences were within 10 CFR 20. Table A (adapted from Table 7.2 Radiological Consequences of Postulated Accidents Final Environmental Statement, October 1973) indicates several of the doses calculated in the FES.

TABLE A

Class	Event	Dose at Site Boundary
1	Trivial incidents	Within 10 CFR 20[1]
2	Small release outside containment	Within 10 CFR 20[1]
3	Radwaste equipment leakage or malfunction	0.046 rem
3	Release of waste gas storage tank contents	0.185 rem
3	Release of liquid waste storage contents	0.005 rem
5	Fuel cladding defects and steam generator leaks[2]	Within 10 CFR 20[1]
5	Off-design transients that induce 0.5% fuel failure and steam generator leak[2]	.001 rem

1. The applicable standard was 0.5 rem whole body or equivalent dose to an organ. Where no specific dose value is listed, releases were expected to be a small fraction of 10 CFR 20 limits for liquid or gaseous effluents.
2. Leakage other than a tube rupture, which was analyzed separately.

Thus, the FES anticipated occasional spills, lifting relief valves on radwaste holdup tanks and fuel cladding defects in the assessment of dose consequences. The FES found that these operational occurrences would not endanger the health and safety of the public because the potential off-site doses were below regulatory limits.

In August 1974, an inspection of the existing radwaste systems found them in compliance with the FSAR and Technical Specifications. The inspection was done prior to the operation of the modified radwaste systems.

A Full-Term Operating License (FTOL) was issued on December 27, 1974. A supplement to the safety evaluation issued with the FTOL noted that radioactive releases for 1970 through 1973 were well within the limits of the plant Technical Specifications. It further noted that augmented effluent treatment systems were expected to be in operation in 1974, which would produce significant improvement in releases, meeting ALARA guidelines. That conclusion was conditional upon the licensee properly operating and maintaining the equipment. The supplemental SER further concluded that the Haddam Neck facility was in conformance with all rules and regulations of the Commission.

The modified radwaste systems were put in operation on July 1, 1975. The NRC found the design acceptable. However, during construction, field changes were made to substitute rupture discs for safety valves on several tanks, such as the waste gas decay tanks and degasifier. The changes were made due to the long lead time to deliver safety valves. No safety evaluation was done for the change by the licensee as required by 10 CFR 50.59, which had been in effect since 1969. The discs ruptured on several occasions before the licensee, with recommendation from NRC, replaced them with safety valves in 1981.

Haddam Neck submitted its final Demonstration of Compliance with 10 CFR 50, Appendix I, on November 1, 1976. The report noted average annual tritium releases were 5761 Ci/yr. It also noted that "uncontaminated drains" were expected to contain liquid with activity about 1% of primary coolant activity. The liquids would be treated prior to release.

An internal NRC memo dated October 14, 1977, contains a detailed evaluation of the radioactive waste systems at Haddam Neck. It concludes that the modified systems were capable of maintaining releases ALARA and within the levels required by Appendix I.

Haddam Neck experienced an unplanned noble gas release in excess of Technical Specification concentration and release rate limits on December 16, 1979. A rupture disc on the degasifier (one of the modified radwaste system components) actuated due to overfilling with water. The overfilling occurred due to failure of the level control relay to stop flow. The dose at the site boundary was calculated at 0.00045 rem. (Comparing this value to the Table A-1 event, "Radwaste equipment leakage or malfunction," it will be seen that the off-site consequences were within the bounds of the FES.) The root cause was attributed to design errors in that a rupture disc was used for pressure relief rather than a safety valve, which would reset once pressure decreased. The root cause analysis did not recognize that the original design specified safety valves, and that rupture discs were substituted during construction.

The licensee considered several actions in response to the 1979 gas release. Two were implemented. The first, PDCR 345, added a liquid level alarm to alert operators that water was collecting in the base of the plant stack. The change was initiated in January 1980 and received its final QA review on September 17, 1982. The second documented action taken was replacing the rupture disc with a safety valve. This was initiated on September 18, 1981, as PDCR No. 413, and given final QA review on September 13, 1982. The design document notes that Haddam Neck took the corrective action in response to an NRC requirement. The requirement was incorporated as an addition to the requirements of NUREG-0578 (Systems Integrity). The design document notes that a total of five unplanned releases in the previous four years had occurred due to rupture disc actuation. Rupture discs were used on the waste gas decay tanks and steam generator blowdown tanks, as well as the degasifier. Some of the discs, not specifically identified, were noted as relieving directly into the PAB or Waste Disposal Building. The building ventilation systems discharged to the plant vent stack, which was a monitored release path. Subsequently, all the rupture discs were replaced.

Radiological Effluent Technical Specifications (RETS) were incorporated in the Haddam Neck Operating License on September 5, 1985, as License Amendment No. 68. The safety evaluation noted the purpose of the proposed technical specifications was to keep releases to the environment ALARA during normal operations and expected operational occurrences. The technical evaluation of the licensee's proposal was done by a contractor whose report is incorporated into the SER. Section 3.1.1 of that report states that "Liquid radioactive wastes are collected in sumps and drains in the various buildings, then transferred to the appropriate tanks in the radwaste building for further treatment." Relative to this description, the NRC team noted that an unplanned, unmonitored liquid release occurred in 1989 when workers processed several drums of containment sump water in the spent fuel building. The processing done in the spent fuel building appears to have been outside the licensing basis of the RETS. The workers treated the water by filtering, and directed the filtrate to the floor drain under the spent fuel pit heat exchanger. The workers believed the drains went to the radwaste system. In fact, the drains led to the yard drains, which allowed the water to leave the RCA via an unmonitored path (See Appendix B for more details).

In 1989, Haddam Neck found 456 fuel pins with throughwall cladding defects during the refueling outage. The defects were caused by machining chips left in the core after thermal shield modifications done during the previous refueling outage. The NRC team noted that the waste gas decay tank accident analysis as described in both the FDSA and UFSAR assumes 1% (320 rods) fuel cladding defects as a design basis. Because the defects observed in 1989 released relatively small amounts of iodine into the reactor coolant system during normal operation, the licensee's fuel monitoring program anticipated only 10 to 12 failed rods prior to refueling. Although the event was reported to the NRC, the licensee did not recognize that the number of defected rods exceeded an accident analysis design basis when the extent of the damage was determined after plant shutdown. However, the actual curie content of the tanks did not exceed 5% of the activity assumed in the accident analysis for purposes of calculating off-site dose consequences.

Conclusions

The original design and safety evaluations anticipated radwaste handling outdoors. As of 1974, the radwaste systems were in compliance with the FDSA and technical specifications. Operational occurrences resulting in spills and releases outside containment were evaluated in the Final Environmental Statement and all were found to be within the regulatory limits for protection of the public.

The design of modifications to the radwaste processing system completed in 1975 met Appendix I requirements. During construction, a field change was made to the design to substitute rupture discs for safety valves to provide pressure relief protection on several tanks. The licensee did not perform a safety evaluation of the change, as required by 10 CFR 50.59. In addition, the field change appears to have met the definition of an unreviewed safety question (USQ), in that a malfunction of a different type than previously evaluated may have been created. If the change was a USQ, prior NRC approval would have been required to make the change. The rupture discs were replaced with safety valves after an unplanned release that occurred in 1979.

Liquid waste processing in the Spent Fuel Building resulted in an unplanned release in 1989. The processing did not conform to the conditions analyzed in the Safety Evaluation Report performed for the 1985 RETS license amendment.

The waste gas decay tank rupture accident analysis assumed 1% (320 rods) defected fuel as a design basis. In 1989, Haddam Neck found 456 defected rods during the Cycle 15 refueling outage. Operation during Cycle 15 appears to have been, in part, outside the design basis for that accident. The fuel monitoring methods used during operation underestimated the number of leaking rods due to the small amount of iodine produced by the defects. After inspecting the fuel and discovering the full extent of the cladding defects, the licensee did not recognize that the 1% design basis for fuel integrity had been exceeded. However, the amount of radioactive gas in this waste gas decay tank was well below the design value used to calculate off-site dose consequences.

5.2 System Operations

Scope

This section reviewed the licensee's procedures and program for the transfer of liquid radioactive material in radioactive waste systems.

Details

The licensee's liquid radioactive waste-handling facilities required transfer of radioactive slurries outside the confines of plant buildings. This practice was not uncommon among nuclear plants licensed in the 1960s, such as Haddam Neck. Haddam Neck's design called for resin liners to be contained in designated pits providing shielding for personnel and dikes for containment of potential spills. Resin liners were stored outside in unroofed areas until 1981, when a spent resin storage facility was built.

The Process Control Program (PCP) for Haddam Neck was proposed in 1979 by the licensee and described the functions of the Liquid Waste System and Purification System. The purpose of the PCP was to ensure that the radioactive waste liquid solidification system was operated to produce a final product that contained no free-standing water and resulted in a completely solidified waste. A PCP is required to ensure that waste is properly characterized as required by 10 CFR 61.56. Liquid radioactive waste that required solidification was processed as directed by approved procedures. The PCP also described the purification system functions, which were to remove impurities from the reactor coolant system during operation or plant shutdown, the volume control tank and RWST, the reactor cavity during refueling and the spent fuel pit water, when necessary. The PCP provided for sluicing of resins to a shipping container in a reinforced concrete shipping cask using demineralized water, which was pumped back to the aerated drain tanks for further processing. The proposed PCP contained details of the process by which concrete was added to radioactive wastes in certain prescribed ratios to form an acceptable waste form for disposal.

A revised Process Control Program for Haddam Neck became effective in 1985 with Amendment 68 incorporating the Radiological Effluent Technical Specifications (RETS) to Appendix A of the operating license. These changes followed the implementation of changes to 10 CFR 20 regarding low-level radioactive wastes and the incorporation of the new 10 CFR 61. The PCP states that Haddam Neck is committed to a management system and procedures necessary to ensure that:

1. all liquid wastes are solidified in accordance with regulatory and disposal site criteria;
2. containers, shipping casks and methods of packaging meet 10 CFR 71 and 49 CFR requirements;
3. waste classification will meet 10 CFR 61 and disposal site requirements, and;
4. approved procedures will include detailed information regarding sample mixing, solidification processes, QA of the solidification process, absence of free liquids, and handling containers if solidification is exothermic.

The stated objective of the Haddam Neck Process Control Program was to ensure safe, effective solidification of radioactive waste liquids and slurries for off-site disposal and to ensure compliance with 10 CFR 71, 10 CFR 61, 10 CFR 20, 49 CFR and disposal site regulations. The details required to meet these commitments were maintained in approved procedures. In 1986, an expanded facility for low-level radioactive waste (LLRW) handling and storage was built.

Subsequent inspections reviewed radioactive waste handling practices and the licensee's PCP. One inspection in 1986 identified four weaknesses in the classification of wastes for Iron-55 and licensee internal audits of the PCP. Corrective actions were implemented and closed out during an inspection in 1987. During an inspection in 1991, the NRC reviewed changes to the radwaste system, including the installation of a spent resin storage tank. The report noted that spent resins were primarily processed by dewatering using a vendor-supplied system. The Process Control Program for both methodologies (e.g. dewatering and cement solidification) were examined by the inspector and determined to be appropriate.

Available radwaste operations procedures controlling the transfer of radioactive slurries to shipping containers revealed that the licensee continued to maintain procedural control over such transfers. The revisions that were reviewed included the following:

"Spent Resin Storage Facility, RPM 3.3-1, Rev.3", 9/19/94
"Set-up of HICs for Resin Slurry, RPM 3.4-2, Rev.4", 5/14/93
"Dewatering of HICs in the Spent Resin Storage Facility,
RPM 3.4-4, Rev.14", 11/22/96
"Spent Resin System Operation, RPM 3.4-6, Rev.8", 12/12/96
"Resin Slurry to Spent Resin System, RPM 3.4-8, Rev.3", 12/12/96
"Shipment of Radioactive Waste Packages, RPM 3.6-1, Rev. 9", 2/11/97
"Set-up and Test of the Chem-Nuclear Set-Up and Test of the Chem-Nuclear Universal Dewatering System, RPM-3.9-8, Rev.3",(Major), 2/15/94

Copies of earlier procedures were not readily available, however, the above procedures included caution statements for control of contamination. The NRC team noted that though there were some incidents regarding radioactive drain transfers, resin spills, cask washdown and contamination from outside storage, these incidents were infrequent. Because these contaminating events occurred outdoors and the boundary of the RCA was close, contamination may have spread to adjacent areas.

Conclusion

Haddam Neck's controls for operating the liquid radioactive waste systems in outdoor locations within the radiological control area appeared adequate over the duration of commercial operation. The licensee maintained approved procedures for radioactive waste-handling operations in accordance with their license requirements. However, some of the outdoor practices may have resulted in the spread of contamination to areas on the licensee's property that were not included in the licensee's survey program. A Process Control Program describing the liquid radioactive waste and purification systems was maintained with appropriate procedural controls. When regulatory requirements changed, the PCP was revised accordingly. Although violations of specific requirements were identified early after the implementation of the revised PCP in 1986, the Process Control Program at Haddam Neck was found to be appropriate.

6. LICENSEE RESPONSE TO IE BULLETIN 80-10

IE BULLETIN NO. 80-10: CONTAMINATION OF NONRADIOACTIVE SYSTEM AND RESULTING POTENTIAL FOR UNMONITORED, UNCONTROLLED RELEASE OF RADIOACTIVITY TO ENVIRONMENT

Scope

This section reviewed the licensee's response and NRC inspection follow-up to actions required by NRC IE Bulletin 80-10. By the end of June 1980, licensees were required to: (1) review their facility design and operation to identify systems that are considered non-radioactive but could possibly become radioactive through interface with radioactive systems (i.e., become contaminated due to leakage, valving errors or other operating conditions); (2) establish a routine sampling or monitoring program for these systems to promptly identify any contaminating events which could lead to unmonitored, uncontrolled liquid or gaseous releases to the environment, including releases to on-site leaching fields or retention ponds; (3) restrict access to contaminated non-radioactive designed systems or evaluate operation in accordance with 10 CFR 50.59, and consider the level of contamination to the radioactive effluent limits of 10 CFR Part 20, RETS and to environmental radiation dose limits of 40 CFR 190; and lastly (4) determine if potential releases comply with requirements for radioactive effluent releases or, if continued operation required a change to technical specifications or constituted an unreviewed safety question, not operate the system as contaminated without prior NRC approval. The Bulletin also stated that if a nonradioactive system was contaminated, decontamination should be performed as soon as possible.

The licensee's original response to IE Bulletin 80-10 was evaluated and found adequate by the NRC staff, but an Inspector Follow-up Item (IFI) was opened due to the licensee's failure to address non-liquid systems in its response. The item was closed in January 1983 (IR 50-213/82-21) by the resident inspectors. The item was opened again in 1990 by a radiation specialist due to positive levels of I-131 detected in vegetation samples close to the site boundary. The IFI was closed again in 1991, when the licensee demonstrated the low safety significance of the non-liquid systems (as addressed in 1982) and the corrective actions to prevent future unmonitored releases. The NRC staff stated in 1991 that the licensee's response was adequate, however, several unmonitored spills and releases outside the radiation controlled area have occurred since the licensee evaluated Bulletin 80-10.

Details

The licensee prepared documentation of all relevant information to IE Bulletin 80-10, including their response. The various dates involved with the licensee's IE Bulletin 80-10 response are listed below:

May 6, 1980:

Issuance of IE Bulletin No. 80-10, titled "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment." The bulletin described a problem at the Brunswick Nuclear facility where the auxiliary boiler was operated for an extended period of time with radioactive water. A tube leak in the firebox of the boiler resulted in an unmonitored, uncontrolled release of radioactivity to the environment. Action items were to be taken by each licensee with a verification letter submitted to the Regional NRC Office.

June 23, 1980:

Licensee submitted response to IE Bulletin 80-10, EN-MO-153. The response stated that actions for Bulletin items 1 and 2 were completed.

February 8, 1982:

Region I radiation specialist reviewed the licensee's documentation of the review performed for IE Bulletin 80-10. The inspector found that the licensee's review did not include non-liquid systems. An inspector follow-up item was opened (IFI 81-11-01). The licensee committed to perform the non-liquid systems review by 11/30/82.

January 3, 1983:

NRC Inspection Report No. 50-213/82-21 documented the licensees actions for non-liquid systems in response to IE Bulletin 80-10. The review was documented in the licensee's report, CYAPCo CN 82-803, dated 11/29/82. The licensee concluded that there was a very low probability that contaminated releases could occur through the non-liquid systems. The follow-up item was closed.

May 17, 1989:

The cover letter for NRC Inspection Report No. 50-213/89-02 stated that the Regional staff was concerned regarding an unmonitored release path that had existed through the Spent Fuel Building floor drains and that the radioactive liquid entered these drains on at least one occasion. The letter stated that the issue of unmonitored release paths was brought to the licensee's attention in IE Bulletin 80-10 and the area warranted further consideration.

June 29, 1990:

NRC Inspection Report No. 50-213/90-11 documented a specialist inspector's review of the licensee's response to IE Bulletin 80-10 and the problem with the clean drain in the Spent Fuel Building that had been used to dispose of contaminated water. The inspector also noted that positive levels of iodine (I-131) found in vegetation samples close to the site boundary could be associated with an unmonitored release path. The inspector requested that the licensee complete the evaluation of non-liquid pathways to close the follow-up item (the inspector did not realize that the item had already been closed in 1983), verify that the remedial action for unmonitored pathways in the original and subsequent evaluations was complete, and review the adequacy of the original engineering evaluation of unmonitored pathways conducted in 1980 (EN-MO-153) in view of the environmental sampling results I-131 in vegetation. The inspector noted that this was an unresolved item (URI 50-213/90-11-01).

January 24, 1991:

NRC Inspection Report No. 50-213/91-01 documented a specialist inspector's follow-up of the URI 50-213/90-11-01. The inspector noted that the I-131 found in vegetation could be explained by previous known releases of noble gases and iodine that were higher than normal releases because of significant fuel cladding defects. The inspector also reviewed the unmonitored release from the Spent Fuel Building to the open trench surrounding the 115 kV switchyard. The inspector noted that the pathway had been identified in the licensee's original review for IE Bulletin 80-10. The licensee had stated the drain was plugged after the pathway was identified in 1980. Sometime between 1980 and 1989, the plug had been removed. As a new corrective action, the licensee plugged the drain line and welded the plug in place to prevent inadvertent removal. The licensee also revised the procedure for monitoring potential pathways to the environment. The revised procedure included plugging and labeling drains, as well as development of a surveillance program to ensure that the pathways are monitored at appropriate frequencies to ensure the systems remain noncontaminated. The inspector closed the item based on the licensee's corrective actions.

The licensee had another unmonitored, uncontrolled release from the Primary Auxiliary Building (PAB) heat exchanger through a drain line to an area drain for the Adams Filter dike in April 1994. The drain emptied into the open trench in the 115 kV switchyard. The total radioactivity released was not significant, but the event cause was attributed to the lack of controls for the drain systems in the radiologically controlled area.

In the period between 1996 and 1997, NRC inspectors questioned releases to the environment which prompted a new review of IE Bulletin 80-10 by the licensee's staff. Two separate contractors were reviewing the potential for non-contaminated systems to become contaminated and the historical information related to past contamination of clean systems. The findings appear to indicate that the initial response to the bulletin by the licensee's staff was a minimal system review and the licensee's program did not require safety evaluations when nonradioactive systems became contaminated. Consequently, several systems that became contaminated did not have a safety evaluation performed. Those systems included the closed cooling water system, the drain systems, the component cooling water system and the turbine building waste water system. For example, the main turbine was known to have contamination from steam generator sludge contaminated by primary to secondary leakage as early as 1970, yet a 10 CFR 50.59 safety evaluation had not been completed for the turbine sump. This was in direct conflict with the guidance in IE Bulletin 80-10. The contractors made recommendations that the licensee is currently reviewing.

The NRC Region I staff inspected the licensee's progress in reviewing the implementation of the guidance from IE Bulletin 80-10. This review is documented in NRC Region I Inspection Report No. 50-213/97-10. The licensee developed and implemented a three-phase program to re-evaluate plant systems relative to NRC Bulletin 80-10 criteria. The program included review of all systems, including current systems in operation and abandoned systems. The licensee performed a comprehensive review of the systems relative to criteria contained in NRC Bulletin 80-10 and was establishing a sampling program to monitor those systems, as appropriate, to ensure detection of potential cross-contamination of normally non-radioactive systems. The review was completed on November 14, 1997. The review of known radioactive systems was for purposes of evaluating system interfaces with typically non-radioactive systems. The review of non-radioactive systems included review of system interfaces and past known contamination history. The licensee developed a safety evaluation status summary for affected or potentially affected systems and was performing safety evaluations for systems considered high priority (i.e., systems known to contain or that had contained radioactive material or had a high potential for contamination.)

The licensee had also established a sampling and analysis matrix for use in evaluating proposed changes to the chemistry sampling program. The licensee revised analysis methods to establish lower limits of detection to meet environmental lower limits of detection. The licensee was also initiating action to review and revise the radiological environmental monitoring program and the off-site dose calculation manual to provide for sampling of alternate release paths (e.g, storm drain system) as appropriate. The potential changes to the off-site dose calculation methods included addition of the external containment sump and RCA yard drain system as a continuous release pathway.

The inspectors noted that although several systems were identified that exhibited low-level contamination (e.g., closed loop cooling, heating and condensate steam component cooling water, turbine sumps) no apparent immediate safety concerns were noted. The licensee had posted the turbine building with information signs indicating the need to contact radiation safety personnel when planning work in the turbine building on a potentially contaminated system.

Conclusions

A recent review by the licensee relative to performance on IEB 80-10 revealed that the initial review was not fully comprehensive because it did not identify all systems that could be potentially contaminated. The recent review also revealed that noncontaminated systems had been used after they were contaminated, and that no safety evaluation had been performed. Very recently, the licensee's implementation and evaluation for continued use of contaminated systems were reviewed by the NRC and was documented in NRC Region I Inspection Report No. 50-213/97-10. The inspectors found that the licensee had established and implemented a task force to reconsider the guidance of NRC Bulletin 80-10, and develop a comprehensive program for decommissioning. This was considered a very good initiative to improve management oversight of station systems that could become cross-contaminated and result in an unmonitored release to the environment.

7. PLANT EXPERIENCE WITH STAINLESS STEEL CLAD FUEL

Scope

The review summarized the licensing and performance history of stainless steel clad fuel at Haddam Neck. The purpose was to determine if the operation of the Haddam Neck plant was within the licensing basis of the fuel design and to identify areas to be considered during site characterization.

The review included examination of documents from the 1967 issuance of the safety evaluation for the Provisional Operating License to the 1994 issuance of License Amendment No. 171, which removed certain restrictions pertaining to stainless steel fuel because the licensee had switched to zircaloy clad fuel. Other reactor systems and components were not examined, except to note an ECCS reanalysis done in 1981 which affected fuel peak clad temperature.

Documents reviewed included pre-operational safety analyses, the Provisional Operating License and amendments, the Full Term Operating License and amendments, the Final Environmental Statement, Facility Description and Safety Analysis (FDSA), Updated Final Safety Analysis Report (UFSAR) and plant design change requests related to radwaste systems. Documents were examined on site at Haddam Neck, in the NRC Region I offices and at NRC headquarters.

A limited scope review was performed of the Haddam Neck power plant's reported releases of solid, liquid and gaseous radioactive material through monitored, unmonitored and uncontrolled pathways. The reviewers compared related NRC regulations and guidance to selected examples of the licensee's effluent and environmental reports from 1979 to 1996.

Background

The defense-in-depth approach establishes four major barriers to isolate fission products from the environment. The fuel is formed into hard, dense, ceramic pellets of uranium oxide which have the capacity to retain a large fraction of the fission products. This provides the first barrier to fission product release. The pellets are sealed inside a metal tube, which is the fuel cladding. A tube filled with pellets is a fuel rod. The cladding forms the second barrier. The fuel rods are arranged in square bundles, held together with several metal plates. The bundled rods are called fuel assemblies. These assemblies are placed in the reactor vessel for use in electricity production. The reactor vessel and associated equipment and piping used to circulate water through the fuel assemblies are the reactor coolant system. This system forms a third barrier to fission product release. The primary system is housed in a metal-lined concrete structure called a containment, which is the fourth major barrier.

As long as any one of the barriers outside the fuel pellets remains intact, fission product releases to the environment can be controlled to levels below regulatory limits for protection of the public. Regulations limit the amount of leakage allowed from each barrier. The leak rates may be directly specified, as with containment structures, or implicit, as in maintaining the fuel in a coolable geometry.

In addition, power plants use active systems to remove radioactive material from the reactor coolant system for processing. The systems collect radioactive materials in various tanks, which hold them for a period, allowing time for decay. Afterwards, the materials are released in a controlled manner or packaged and shipped for disposal.

The review that follows discusses fuel defects that formed throughwall penetrations of the cladding, no matter how small, that allowed fission products to enter the reactor coolant system.

Overview

Haddam Neck was one of the few commercial nuclear plants to use stainless steel fuel cladding. The NRC and its predecessor, AEC, analyzed the fuel performance on several occasions and found that the design met regulatory requirements. The Electric Power Research Institute (EPRI) examined the performance record of over 550,000 stainless steel clad fuel rods in the United States and Europe in a report published in 1982. EPRI concluded that stainless steel clad fuel performance was excellent overall, with the exception of certain specific cases, such as the Haddam Neck cladding defects in 1979.

In 1992, Haddam Neck began a conversion to zircaloy cladding to reduce fuel costs. The conversion was completed in 1995.

Haddam Neck experienced throughwall fuel cladding defects in the range of 45 to 456 rods in Cycles 8, 15, and 16, which occurred in 1979, 1989 and 1991, respectively. The Haddam Neck reactor vessel contained 157 fuel assemblies, with a total of 32,028 fuel rods. Fuel performance and licensee actions during those cycles were examined to determine the extent to which the licensee’s action conformed with the licensing basis.

7.1 AEC/NRC Evaluations

In the first seven years of plant operation, the stainless steel clad fuel used at Haddam Neck received three major evaluations from AEC and NRC staff. A fourth major evaluation was performed by the NRC in 1983. This last evaluation was issued in License Amendment No. 52, as noted in Section 7.2, to approve a new fuel design. The first safety evaluation was published in May 1967 to support the Provisional Operating License. It was noted that the fuel design was similar to other operating plants using stainless steel. The staff concluded that operation within the Technical Specifications would not cause cladding defects in excess of the design basis (1% of the fuel rods). The second evaluation, in July 1971, was done to support issuance of a Full-Term Operating License (FTOL). It noted that upgrades to the ECCS reduced the calculated peak clad temperature (PCT) by 50°F. A third evaluation was published in December 1974. The FTOL issuance had been delayed to prepare an environmental impact statement in accordance with National Environmental Protection Act. Due to the delay, the FTOL safety evaluation was updated. The staff noted that Haddam Neck fuel performance was bounded by conditions at San Onofre Unit 1, and concluded that the likelihood of clad collapse was remote. A re-analysis of peak cladding temperature by the licensee using an updated Westinghouse model calculated the PCT as 2300°F. The staff concluded the plant met the Interim Acceptance Criteria for ECCS.

7.2 Defects in Cycle 8, 1979

Elevated reactor coolant iodine levels at the end of Haddam Neck's Cycle 8 in 1979 indicated fuel cladding defects. The fuel inspection during refueling revealed 36 leaking assemblies, containing about 45 leaking rods. All the assemblies came from one batch, Batch 8, which used BNFL-supplied pellets and Babcock & Wilcox (B&W) fuel fabrication services. Batch 8 assemblies had the highest burnup (29,000 to 36,000 MWD/MTU) in the core. The faulted assemblies were removed from service. The fuel pellet supplier, British Nuclear Fuels Ltd. (BNFL) suggested that a power ramp at the end of Cycle 7 may have initiated the defects. Power ascension restrictions were put in place until the cause of the defects was determined.

For Cycle 9, six assemblies of Batch 9, which had seen service in Cycle 8, were loaded into the core. These assemblies had an average burnup of 24,200 MWD/MTU prior to loading. They contained 162 rods with pellets made by BNFL, out of a total of 1062 fuel rods. During Cycle 9, iodine indicated some leaking assemblies. Fuel sipping after Cycle 9 found 8 or 9 leaking assemblies in Batch 9. They were taken out of service.

In November 1981, the licensee forwarded a final report on the cause of the 1979 fuel defects. The investigation was done by Battelle, Columbus Laboratories under an EPRI contract. They concluded that the following elements played a role in the failures: 1) fuel pellet chips caused high localized stresses in the cladding, and 2) the lower propensity of Batch 8 fuel to densify led to enhancement of fuel-clad contact pressure. A power change near the end of Cycle 7 may have played a role in causing the defects. Changes were made to the fabrication process to avoid pellet chipping, and refinements to the fuel design were planned for future batches, primarily an increase in the fuel-clad gap.

Haddam Neck submitted an amendment request to change the Technical Specifications to allow use of a revised fuel design developed to avoid fuel defects from the mechanism discovered in Cycle 8. A change to address concerns over operating with actual reactor core inlet temperature below its design value was included in the request.

License Amendment No. 52 was issued March 3, 1983, to revise the Technical Specifications to reduce the maximum allowable linear heat generation rate (LHGR) and adjust the axial power vs. offset curves accordingly. The changes were needed to allow use of a revised fuel design, developed to avoid the clad defect mechanisms observed in 1979. As discussed in Section 7.3 below, additional reduction of LHGR was imposed due to reactor operation with a lower than design core inlet temperature, which affected the calculated peak cladding temperature (PCT).

NRC's Safety Evaluation performed audit calculations to confirm the fuel design results submitted by Haddam Neck. The revised fuel design, run at reduced peak power levels, was found to be bounded by conditions previously analyzed and acceptable.

The changes were effective in minimizing fuel damage. Subsequently, fuel used during Cycle 9, 10 and 11 experienced progressively fewer clad defects, as indicated by reactor coolant iodine monitoring.

7.3 ECCS Performance

Haddam Neck had assumed that operation of the reactor with lower than design core inlet temperature was conservative with regard to the emergency core cooling system (ECCS) performance analysis. The licensee believed that peak clad temperature would decrease in a design basis accident if the core inlet temperature was decreased. This was not the case when the analysis was performed. The erroneous assumption had been accepted since Cycle 1, when core inlet temperature was reduced from its design value. The error was reported on December 11, 1981. However, the licensee did not provide the date of the temperature reduction.

As a corrective action to the reported error, the licensee proposed Technical Specification revisions to assure adequate ECCS performance as part of License Amendment No. 52. The ECCS analysis could meet PCT limits if certain conservative model assumptions were relaxed. However, the licensee found that the ECCS analysis would meet the Interim Acceptance Criteria limiting PCT to 2300°F, without relaxing the conservative model assumptions, by reducing LHGR. The staff accepted the analysis with the reduced LHGR.

7.4 EPRI Evaluation

In 1982, EPRI published an evaluation of stainless steel cladding for use in LWRs. It examined the performance of more than 550,000 stainless clad fuel rods used in six commercial power reactors located in the United States and Europe. Stainless steel clad for BWR fuel was inferior to zircaloy. In PWRs, however, stainless steel performance was comparable or superior to zircaloy.

EPRI found that stainless steel cladding had been widely used in the early years of nuclear power in a variety of facilities, such as power reactors, test reactors and ship reactors. However, zircaloy cladding provided better neutron economy and thus lower fuel costs. At the time of the report, only LaCrosse, Haddam Neck and San Onofre 1 continued to use stainless steel clad fuel in the U.S.

The report found that the performance of stainless and zircaloy fuel in normal conditions was similar. The response of the two materials to transients differed, depending on the transient considered. However, both materials were considered acceptable. Stainless steel had much higher permeability to tritium, which was reflected in higher tritium releases from plants that used stainless steel. EPRI concluded that the tritium release from the three U.S. LWR plants using it at the time of the report was not an environmental problem. Zircaloy had lower thermal neutron absorption, making it more economical since lower enrichment fuel could be used.

Of the six reactors examined for stainless steel fuel performance, three reported no defects. One reported two collapsed fuel rods but no other defects. Two, which include Haddam Neck's 1979 experience, reported incidents of approximately 50 to 100 leaking rods during a cycle but otherwise no defects. Fuel inspections were not always extensive. Nevertheless, reactor coolant iodine data support the assertion that very few rods stainless steel rods had cladding defects during operation of the plants, other than the incidents noted.

EPRI concluded that the performance record of stainless steel clad fuel was excellent. The performance was considered more significant because most of it had been achieved without any power maneuvering restrictions. The favorable results were attributed to the lower linear heat generation rate of PWR stainless steel fuel compared to zircaloy clad fuel.

7.5 Defects in Cycle 15, 1989

Cycle 15 began March 3, 1986 and ended on September 2, 1989. The operating cycle was followed by a 346-day refueling outage.

Reactor vessel internals work and fuel inspection and repairs accounted for the length of the outage. Fuel damage had been caused by machining debris left in the core after doing thermal shield work following Cycle 14. A number of metal chips got caught in the fuel, primarily at the bottom plate. Coolant flow caused the chips to rub against the fuel clad resulting in debris-induced fretting defects.

Cycle 15 experienced throughwall fuel cladding defects to 456 rods. Approximately 1500 additional rods sustained defects greater than 20% throughwall. Identification of the damage was complicated by the relative insensitivity of ultrasonic testing for detecting defects located at the bottom of the fuel rod. Additional testing methods were used to verify clad condition.

On December 15, 1989, the licensee reported 281 rods with throughwall defects in LER 50-213/89-20 on the basis of serious degradation of a principal safety barrier. The report may have been filed late, since documents show the licensee was aware of the damage as of October 19, 1989. No followup reports were filed as the extent of damage grew larger, eventually reaching a total of 456 failed rods. The NRC historical review team noted that the design basis for the waste gas decay tank (WGDT) rupture accident was 1% failed fuel, or 320 rods. No record has been located to demonstrate that the licensee recognized that this design basis had been exceeded. However, the design basis for the WGDT rupture accident also specifies the maximum amount of radioactive gases available in the event of a release. The licensee reported that the maximum curie content of the tanks over the period 1988 through 1989 was less than 5% of the design basis value.

The licensee had a fuel evaluation program in place during Cycle 15, but the evaluation method was not suitable for quantifying the number of failures due to the unique nature of the defects. The defects in the stainless steel clad occurred at the bottom of the rods, which limited the movement of water in and out of the rod due to gases trapped above the throughwall penetration. Water would enter the rod until the rod interior gas pressure equalized with reactor coolant pressure. The water remained in place unless a pressure or temperature change occurred. This resulted in significantly lower amounts of iodine in the reactor coolant than was usually observed when fuel cladding was breached. Noble gas concentration was considerably higher than usual, but this parameter was not used in PWR fuel evaluation procedures at the time, either at Haddam Neck or in the industry. The evaluation method used during Cycle 15 used iodine as the indicator and predicted 8 to 12 defective rods.

In response to the defects, the licensee conducted an extensive fuel inspection. Considerable effort was given to cleaning debris from the fuel and core, reconstituting fuel assemblies and improving the fuel monitoring program. In addition, the thermal shield was removed due to degradation of its support system.

The licensee devised a model to quantify throughwall defects during operation, which would indicate defects caused by the debris-induced fretting mechanism, by correlating Cycle 15 reactor coolant iodine and xenon measurements with the observed number of defects. Including xenon in the evaluation improved the accuracy of the estimated number of defects. The licensee presented calculations suggesting the method yielded defect estimates from 0% to 22% higher than the actual value.

7.6 Defects in Cycle 16, 1991

Cycle 16 began August 15, 1990, and ended October 17, 1991. The refueling outage lasted 149 days.

Because a certain amount of debris was expected to remain after the cleaning, and about 65 rods were expected to leak during Cycle 16, the licensee proposed amendments to its Technical Specifications that would limit the number of defects to 160 rods. This value was selected based on a steam generator tube rupture accident and was consistent with the Technical Specification limit of 1.0 µCi/g Dose Equivalent Iodine (DEI) activity. An action statement was included that required placing the reactor in hot shutdown if the estimated number of defected rods exceeded 160 rods for seven consecutive days. The proposed technical specification also included surveillance requirements to monitor the number of defected rods. A new basis statement was added which stated that a correlation method was the means to implement the surveillance requirement.

The proposed Technical Specification was issued on January 4, 1991, as license amendment No. 134. Prior to issuance, the licensee implemented the requirements through administrative procedures.

In March 1991, the plant went to Mode 5, cold shutdown, due to inoperable containment air recirculation fans. Reactor coolant iodine spiked to 1.78 µCi/ml, which exceeded reactor coolant specific activity limits. The only required action in Mode 5 was to increase the frequency of sampling until the DEI decreased below the limit. The event was reported in the annual report as required by the technical specifications. Prior to the shutdown, the fuel monitoring program estimated 25 defected pins. After the startup, the defected pin estimate spiked to 130 rods, then decreased to 30 defected pins.

On August 19, 1991, the plant began a shutdown as a precaution against the approach of Hurricane Bob. By the time power had been reduced to 40%, it was clear the hurricane would bypass the site, and the plant returned to full power. The xenon spike that occurred after the maneuver caused the indicated number of throughwall fuel rod defects to increase to 418, although I-131 did not exceed 0.01 µCi/ml. The licensee projected that it would exceed the seven-day LCO, and met with NRC staff on August 26, 1991, to discuss the issue. Haddam Neck personnel presented evidence that the spike was similar to others observed during power maneuvers, and that an alternate estimation method, based on those examples, could be used to better evaluate the spike on August 19. CYAPCo asserted that the LCO did not apply because alternate estimation methods yielded lower numbers. The NRC staff did not object to that assertion.

The Haddam Neck control room log recorded exiting the LCO on August 28, 1991, within the allowed seven-day period, on the basis of a plant chemistry report that the defective fuel estimate decreased to 95 rods using an alternate estimation method. However, the defective fuel estimate based on the method approved in the Safety Evaluation for the applicable Technical Specification did not decrease below 160 rods until September 1, 1991, about 10.5 days after the first indication that the LCO had been entered.

CYAPCo performed a safety evaluation of the change to their defected fuel estimation procedure prior to applying it to the surveillance. The revised procedure used a graphical method to plot the number of fuel clad defects projected to exist in ten days. It allowed the alternate method to be continued for ten days before concluding that the number of defects had changed. The plant staff concluded that no unreviewed safety question was involved and no change was required to the Technical Specifications. That may have been erroneous. The revised method appears to have been less conservative than the method specified by Technical Specifications since it yielded a lower value. Thus, the change may have reduced the margin of safety, which fits the definition of an unreviewed safety question. In addition, the basis of the Technical Specification described the surveillance method to be used to comply with the requirement. Changing the method may have been a change to the Technical Specification basis.

A fuel inspection done after Cycle 16 estimated that 102 rods were defective. It is not clear from available records if the inspection examined all the fuel assemblies. However, the revised fuel performance program indicated about 100 defected rods at shutdown, which agreed with the number of defects found.

Starting with Cycle 17, the licensee began changing to zircaloy clad fuel. The conversion was complete, except for 5 assemblies, by Cycle 19.

The licensee's Semiannual Radioactive Effluent Release Reports that reported the effluent impact from these fuel defects included a summary of the quantities of radioactive liquid and gaseous waste effluents, including any unplanned or abnormal releases, a summary of meteorological data associated with the gaseous effluents, an assessment of the radiation doses from the radioactive liquid and gaseous effluents released to the environment, quantities of radioactive waste disposed of and changes to the Radiological Effluent Monitoring and Offsite Dose Calculation Manual. The Semiannual Radioactive Effluent Release Reports contained plant data in accordance with the guidance in NRC Regulatory Guide 1.21, Revision 1, June 1974. The reported radioactive effluent releases were within the quantities projected in the Final Environmental Statement, which was issued by the Atomic Energy Commission in October 1973. The reported effluents and the associated calculated annual doses to a member of the public were in accordance with the ALARA criteria of Appendix I to 10 CFR Part 50. The licensee used accepted NRC methodology contained in Regulatory Guide 1.109 for the calculation of annual doses to man from the release of radioactive material in nuclear power reactor effluents.

These reports noted that the solid waste streams had transuranic isotopes, which resulted in primary side spent resins often being classified as Class C. (Most power reactor waste streams are typically Class B waste.) However, the effluent and environmental releases were not impacted from the fuel defects, except for one quarterly Technical Specification limit following the 1989 fuel failure. Annual exposure limits were not exceeded.

Conclusions

Haddam Neck stainless steel fuel received four major evaluations from AEC and NRC between 1967 and 1983. The design was acceptable on each occasion and was bounded by conditions at the San Onofre Unit 1 reactor.

The Cycle 8 (1979) fuel defects were due to manufacturing defects, exacerbated by a power ramp performed near the end of Cycle 7. Power ascension limits combined with fuel design changes were effective in minimizing fuel cladding defect formation due to manufacturing defects.

More than 1% of the fuel rods had defects at the end of the Cycle 15 (1989). This value exceeded the design basis value for fuel rod defects found in the FSAR analysis of the waste gas decay tank rupture accident. However, the actual curie content of the tanks was less than 5% of the design basis value assumed for calculating off-site dose consequences.

The licensee's safety evaluation of the change to the failed fuel estimation procedure used in Cycle 16 (1991) may have been in error when it concluded that no unreviewed safety question existed and no change was needed to the technical specifications. If either condition existed, prior NRC approval would have been required to make the change.

The results from operating with defected fuel were a gaseous release exceeding the quarterly RETS limit in 1989. Also in 1989, positive levels of I-131 were detected in vegetation samples taken near the site boundary. The 1989 release did not exceed 10 CFR Part 20 exposure limits. Overall, the liquid and gaseous radioactive waste effluent data were properly documented and reported in accordance with 10 CFR 50.36a and Criterion 60 of Appendix A to 10 CFR Part 50. No significant errors or omissions were identified. The calculated annual doses were in accordance with the ALARA criteria of Appendix I to 10 CFR Part 50. Another result of the fuel defects was alpha contamination of the interior surfaces of plant equipment.

Concerns to be addressed during site characterization include characterization of alpha contamination of plant primary and secondary systems to determine appropriate procedures for dismantlement and worker protection. Although environmental data do not indicate significant transurancies (indicated by Am-241 gamma), characterization of the site should take the potential for alpha contamination into account.